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memorandum

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SUBJECT: Total Neutron Production from Photo-neutron Reactions

The total neutron production as a function of energy from photo-neutron reactions has recently become of interest to a number of people. Currently, the data team has available total neutron production data from Berman et al.[1-2] In an effort to make it easier for the typical user to generate the appropriate data for their specific materials, I have written a couple of simple routines to massage the data into the appropriate format, and store them on CFS along with the data itself (on both open and secure). An index to the available data is stored on CFS as /x6data/photonuc/index.txt , where the filename for each data file is given in the last column. The last section of this memorandum details how one may create a data file for a specific material of interest.

Original Berman Data

The original Berman data is given as the total neutron production from photo-neutron reactions in millibarns as a function of energy in MeV. The total neutron production is calculated in the following manner:

$$\text{Total-N} = (\gamma,n) + 2*(\gamma,2n) + 3*(\gamma,3n) + \ldots + \text{nubar}*(\gamma,\text{fission})$$

Figures 1 and 2 show the data for the individual reactions plus the total neutron production for ^{235}U and ^{238}U, while Figure 3 shows a comparison of the total neutron production for the two nuclides. The total neutron production data from Berman is stored on CFS under /x6data/photonuc/eval with the corresponding filename from the index. No energy or angle distribution data is currently available, only the total number produced.

Conversion of Berman Data to a Standardized Energy Grid

The Berman data was then converted to a standardized energy grid, and the cross section units converted from millibarns to barns. The standardized energy grid has 291 values extending from 1.0 to 30.0 MeV in steps of 0.1 MeV. The following three assumptions were made during this conversion process:

* The last cross section value was assumed for all energies greater than the final energy of the original data. The first cross section value was assumed for all energies less than the initial energy given on the original data file.
* Linear-Linear interpolation was used between data points.
* When more than one data file has been available for a nuclide, I used the most accurate experimental data (this is not always the most recent experimental data by date). No attempt has been made to combine data from different experiments.

These assumptions provide for the most conservative estimate of the total neutron production from a safety point-of-view, by slightly overestimating the actual total neutron production. The data file for each nuclide is stored by name on CFS under /x6data/photonuc (ganat, b10, c12, ...). A selection of files have been put on CFS for your use. Other data listed in the index can be made available for your use relatively easily. Please contact the Data Team if you need data for other nuclides and give the appropriate filename(s) from the index.

Calculation of the Total Neutron Production for a Specific Material

A simple program called addmat.f was written and is stored on CFS as /x6data/photonuc/addmat.f . This program interactively queries the user for an output filename, and the input filenames and atomic fractions for the nuclides that make up the material of interest. As example is stored under the /x6data/photonuc directory as test_in and test, the input and output information for a simple test case. For this test case, the user will see the following on their screen where the program queries are noted in **bold-face type** followed by the user's response:

```
>addmat
 How many nuclides are in material
4
 Input the name for the output file
test
 Input file name and atomic fraction
   for each nuclide
b10 0.25
c12 0.25
u235 0.25
pu239 0.25
```

The user can run this simple test case using a different output filename and compare to 'test' to ensure that they are running the program properly. Be sure to use the atomic fraction and not the weight fraction in producing the data for your specific material. The output file for your material will have labels identifying the columns and appropriate units. A typical use of this kind of data with the FM card is discussed in the MCNP4A manual, Chapter 2 page 2-83 and Chapter 3 pg. 3-72. Care must be taken to ensure the proper unit conversions and volumes are used in the calculation.

[1] "Atlas of Photoneutron Cross Sections obtained with Monoenergetic Photons", B. L. Berman, LLNL report, UCRL-78482 (1976).

[2] "Photofission and Photoneutron Cross Sections and Photofission Neutron Multiplicites fro 233U, 234U, 237Np, and 239Pu", B. L. Berman et al., Phys. Rev. C, **34** p. 2201-2214 (1986). (and other similar articles).

Figure 1: Photo-neutron Cross Sections and Total Neutron Production for U-235

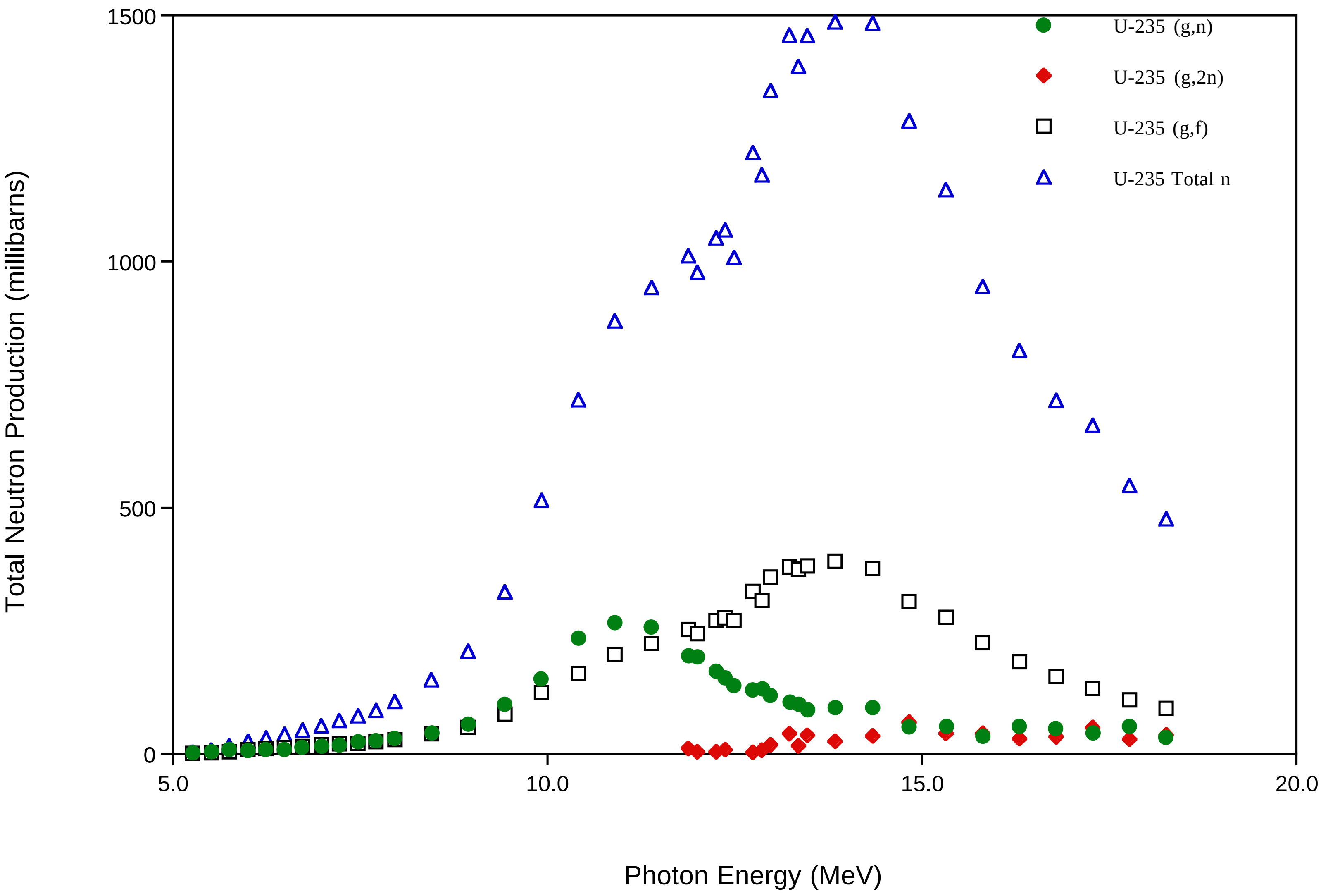
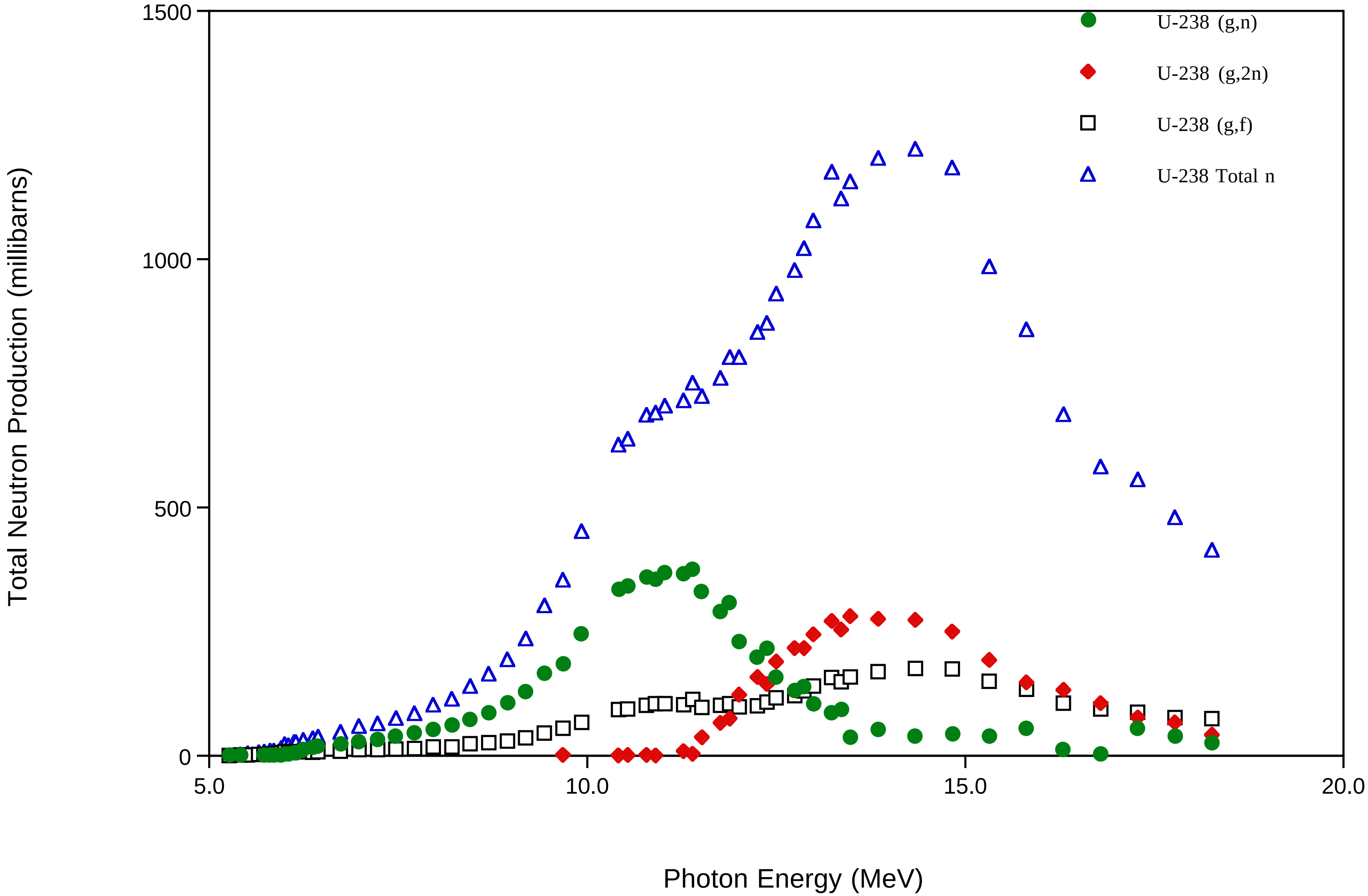


Figure 2: Photo-neutron Cross Sections and Total Neutron Production for U-238



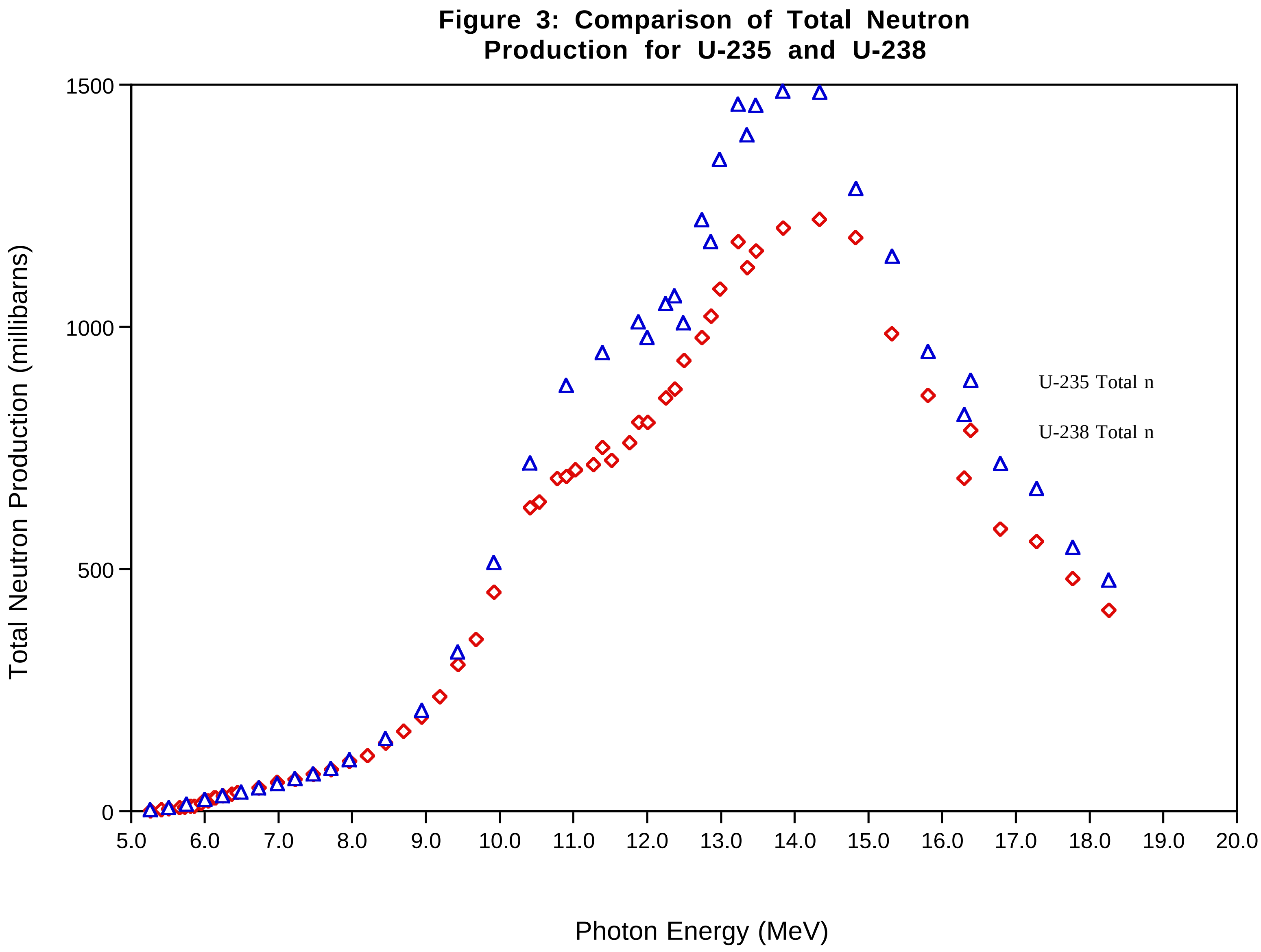


Figure 3: Comparison of Total Neutron
Production for U-235 and U-238
Total Neutron Production (millibarns)
1500
1000
500
0
5.0
6.0
7.0
8.0
9.0
10.0
11.0
12.0
13.0
14.0
15.0
16.0
17.0
18.0
19.0
20.0
Photon Energy (MeV)
U-235 Total n
U-238 Total n

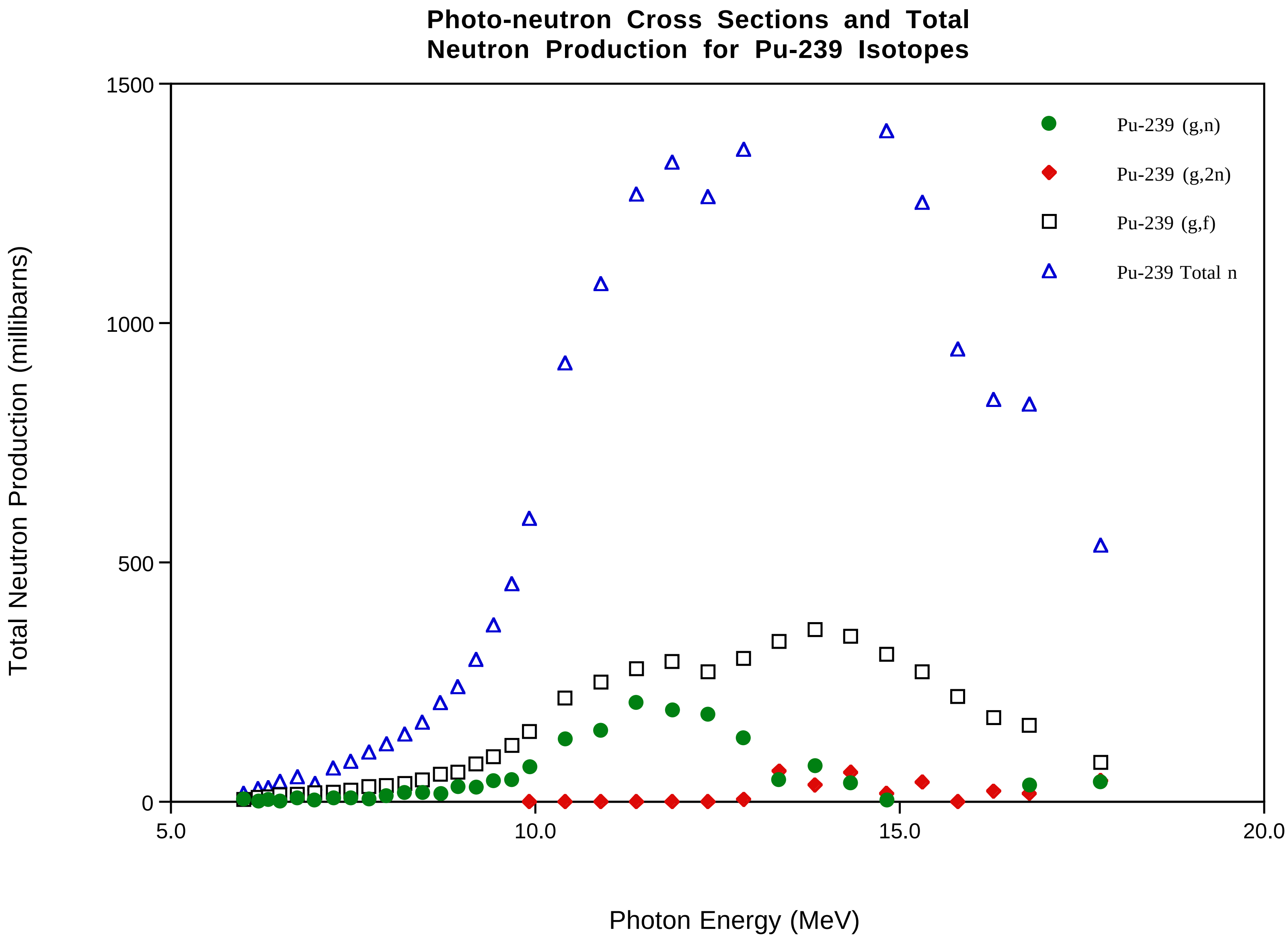


Photo-neutron Cross Sections and Total
Neutron Production for Pu-239 Isotopes
Total Neutron Production (millibarns)
Photon Energy (MeV)
1500
1000
500
0
5.0
10.0
15.0
20.0
Pu-239 (g,n)
Pu-239 (g,2n)
Pu-239 (g,f)
Pu-239 Total n